Pressurized Water Reactor (PWR) System Simulation and Disturbance Analysis for Anomalous Transients and Degraded System Conditions

V. K. Dhir, S. Guarro, J. C. Lin, M. Motamed and D. Okrent
University of California
Los Angeles, California 90024

Abstract

In this paper potential applications of disturbance analysis to improve availability and safety of light water reactors (LWR's) are discussed. Needs for developing on-line computer aided guidance to the reactor operator during anomalous transients are pointed out. Currently available methods to simulate primary and secondary systems of a pressurized water reactor (PWR) during anomalous transients and other conditions severely degraded from normal operation are reviewed. Limitations of these methods for simulation of operational transients are mentioned. Finally, using one of the existing codes, steam line rupture in a PWR is analyzed.

INTRODUCTION

In the past considerable effort has been made to model as well as to experimentally understand various phenomena that may occur during a hypothetical loss of coolant accident (triggered by a large break in the primary coolant pipe line) in a light water reactor (LWR). However, recent events at the Three Mile Island-2 nuclear power plant have shown that to help guide a reactor operator, capability of on-line computer aided analysis of anomalous transients or disturbances is of great importance. At present our knowledge of this area of reactor system behavior is limited. The purpose of this paper is to discuss the existing methods for analyzing disturbances and operational transients in nuclear power plants.

The term "disturbance analysis" has been formulated in the recent years to denote a certain type of computer implemented systems which operate on a large and complex process (such as a nuclear power plant) can provide in real time vital information concerning the operating status of the plant systems and components. The main task of such a system is to detect the onset of any perturbations with respect to the process normal conditions, to identify the prime cause of such perturbations, and finally to suggest, before the plant operability is compromised, the best actions to be taken by the human operators to stop their propagation.

Historically the first attempt to develop a diagnostic tool of this kind must be backdated to the mid 60's when a computerized logic system was devised and installed at the United Kingdom's Oldbury nuclear power station, with the purpose of interpreting and giving information to the plant operators about the interrelations between the plant activated alarms. This first attempt was less successful than expected, mainly because of the insufficient capabilities of the computer hardware available at that time. But starting a few years ago the original idea was again undertaken in a joint effort by the OECD Halden Reactor Project (based in Norway) and the German "Gesellschaft für Reaktorsicherheit" based in Munich, Germany. A preliminary version of the resulting new "DAS" (Disturbance Analysis System) has been tested at the Halden research reactor site, and an improved and more complex version is being installed at the 1300 MWe PWR at Grafenrheinfeld, Germany, for extensive testing which should last for about one year.

In the U.S., the Electric Power Research Institute [1] has started a similar project in 1976 and its system has been undergoing tests on a CE PWR plant simulator.

The EPRI [1] and German methodologies can be considered equivalent in their essential features. In the next section a functional description will be given of how a typical DAS "works", for the purpose of offering a basis for discussion of the theoretical capabilities of such a system and the problems which must be solved in order to obtain a successful implementation of those capabilities in practice.

Anomalous transients are the type of transients initiated by an anticipated failure (initiating event), during the operation of a reactor. Generally, this will include events termed in the ANS classification as "conditions II" and "conditions III". Some of the anticipated failures in a

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PWR which fall in these categories are: loss of feedwater pumps, control system malfunctions, loss of coolant accidents (small breaks), etc. The reactor protection systems alone are usually sufficient to maintain plant safety for condition II (one/reactor year) events, while in condition III (one/100 reactor years) events interaction of Engineered Safety Feature (ESF's) is required. Reactor safety study (WASH-1400), considers these failures as important factors in risk assessment. However, the probability of an anticipated failure leading to a potential accident which could cause large scale radioactive release, or seriously damage the plant, is believed to be very low [2]. The probabilities of an anticipated transient which leads to an accident with serious consequences may be higher than presently believed, if multifailures of components such as the standby systems and active systems were given higher probabilities of occurrence due to human error or common mode failure. Operator action is normally expected to have a favorable effect on mitigation of the consequences of the accident, however, under certain conditions, the operator, due to misjudgment, may take an action which is unwarranted.

DISTURBANCE ANALYSIS SYSTEM (DAS)

It is opportune to give first here a very simple example of application of the DAS methodology, to clarify what such a system can typically do for us. Suppose we have a certain flowrate through a heat exchanger, which is kept nearly constant by an automatic control loop (composed of a flow transmitter, controller, and valve with relative actuator); suppose also that at a certain time the flowrate falls unexpectedly below the desired set-point: this could be the result of different original causes, such as a leak, or a transmitter failure, or a controller failure, or an actuator failure. The DAS will be able to detect immediately the onset of the abnormal situation, then single out the actual cause among all the possible ones, and finally suggest a counteraction to obviate the undesired situation: for instance, making use of a CRT display, it will first issue a message of the following kind: "disturbance in progress: flowrate in loop xxx falling below lower limit"; and then, in the term of a few seconds and upon identification of the failure, for instance, in the loop controller, another message, such as: "flow controller FCxxx failure", together with a suggestion for the action to be taken as a proper countermeasure: "switch flowrate control in loop xxx to manual".

In order to obtain the diagnostic and "suggestion" messages to be displayed, as in the example above, to the process operators, the DAS will have to perform several functional steps, having as the essential part the disturbance identification, that we will try to describe below.

The first step is to obtain from the process instrumentation what can be called the "disturbance pattern", that is a univocal, but sufficiently simple description of the process, given at suitable time intervals (typically 1 to 5 seconds in the present applications). This will contain enough information regarding the operating conditions of the plant systems and components. This is presently done by assigning to every measured process variable a "high" and a "low" limit, so that generally speaking every variable is for the DAS described as a three valued parameter (high, normal or low). In this way a computer generated pattern can be obtained, prividing the DAS with a simplified image of the process, and therefore of any disturbance which might be present in it (of course if all the variables are in the "normal" range no disturbance is considered to be present).

In the case in which a disturbance is detected (one or more variables being out of the normal range) the next step will be to match the above "disturbance pattern" against a suitable model of the plant process, to identify the primary causes which are producing the present upset conditions and the path of propagation of the disturbance to other plant systems and components. As the analysis progresses, diagnostic messages and suggestions will be displayed, as we saw in the example given above, to provide the plant operators with information about the disturbance origin and progress, and, even more important, to help him find a way to stop or limit its propagation. From what we have already said it appears that the process model that is needed to perform this second step must contain all the possible information about both the known failure or malfunction modes of the plant systems and/or components and the cause-effect mechanisms linking these modes to one another. It seems therefore natural that the form which has been chosen in the existing DAS's for this model is the so called cause-consequence diagram, in practice a type of fault tree in which the logical relations between the tree branches have also a meaning of temporal sequence (with time delays eventually inserted in the tree to provide a description of the time frame of the cause-consequence relations). Clearly, the logic variables appearing in the cause-consequence model of the process are chosen in such a way that it will be possible in all conditions to establish a direct correspondence between them and the process image (or "disturbance pattern") constituted by the three valued variables mentioned before. When such a correspondence is fully established by the computer program performing the analysis, including also the particular values that the variables in the process image have at that particular time, the disturbance at its current stage of propagation can be considered as identified, and the next functional step will be undertaken and completed by the DAS. That will be to establish another suitable correspondence between the present identified stage of propagation of the disturbance and the messages which are to be displayed to the human operators (recall once more the simple exemplification given at the beginning of the section).

As an illustration of how the DAS methodology works consider in Figure 1a the schematic diagram of a steam generator in which we have assumed that the valve position, V, and the primary temperature, T_1, are the only parameters that can vary spontaneously, but the primary mass flowrate, m_1, remains constant. We restrict ourselves to this artificial situation since we are only interested here in presenting a very simple example of how the cause-consequence structures can be developed and used.

For the top event "level 1, low" the logic structure derived for our system will be the one given in Figure 1b. Two possible applications of DAS are given below.

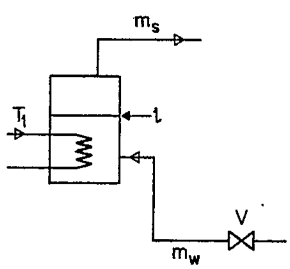
Illustration 1

Suppose that a) is the only observable node and that it becomes activated at time 0. We will have:

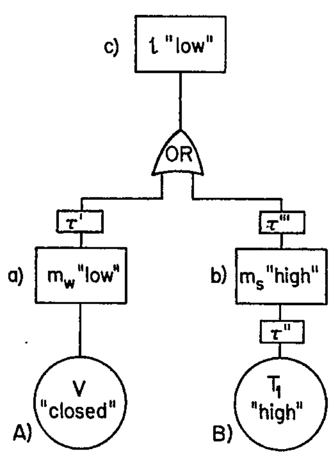
- prediction: "1 low in τ' seconds"
- diagnosis: "valve V closed accidentally"
- suggestion message: "open valve V manually"

Illustration 2

Suppose that b) and c) are the only observable nodes and that c) becomes activated at time 0 but b) does not. No prediction would be obviously available in this case but the diagnosis and suggestion would be exactly as before, since we can exclude the disturbance path B) → b) → c) for which b) would have been activated before c). Of course, if we had considered the possibility of instrumentation failure (as for instance a faulty m_s high reading) the above conclusion could not have been drawn.



a. Schematic Diagram of a Steam Generator System



b. Logic Structure

Fig.1 A Simple Logic Structure for DAS

Use of DAS to Improve Plant Availability and its Potential Application to Plant Safety

The projects carried out so far in the area of the DAS's, as applied to nuclear power plants, have all been basically oriented in the direction of improving the plant availability.

The approach used to develop the cause-consequence models of the plant of which we discussed in the previous section is rather simple at the conceptual level. Typically it starts from a statistical study to identify the most frequent plant perturbations that would normally bring the plant to a forced shutdown if a counteraction is not undertaken and successfully completed. Having identified those, the attention of the DAS designer is focused only upon the ones of them that develop in a time frame lending itself to a DAS type of elaboration (typically 20 seconds or more). Finally the cause-consequence diagram models are prepared, with the explicit purpose of including in them enough details to give full description of all the selected process perturbations, so that the same will be properly identified using the information provided by the plant instrumentation and according to the functional steps that we have described in the previous section.

Nothing in principle forbids one to extend the application of the DAS methodology to the identification of situations affecting, rather than the availbility of a nuclear power station, the very safety of its operating conditions. Again it would be desirable if this new type of application included, if possible, suggestion of the actions to be taken to bring these situations back to normality. Unfortunately the problem becomes here rather complex. In fact, in a DAS application meant only to improve plant availability, the designer could limit himself to dealing with a pretty well defined number of disturbances which would have, as a whole, a statistically high incidence in determining plant outages. So, if the DAS is made able to diagnose those disturbances, the designer will be sure of having reduced drastically the potential for plant outages, which is the design goal. In the EPRI DAS this concept is actually brought to the extreme consequences: there, in fact, the logic models, of which we have already explained the functional role in the system, are derived in such a way as to model a list of preidentified disturbances which are judged relevant to the designer's purposes, rather than try to model the plant itself with all the known failure/malfunction modes of its components: in other words only the component failure or malfunction modes associated with the listed disturbances will be modeled in the cause-consequence diagrams, all the other ones will be neglected.

In a safety type of application a similar type of approach doesn't seem to be viable. In fact, we will now be dealing essentially with the so called "rare events", and therefore no real statistical support will be available anymore to identify beforehand any "list" of disturbances to be considered more relevant, according to whatever criterion we would like to follow (to say nothing of the fact that to establish such a "relevance criterion" would be all but easy). Recent experience (Browns

Ferry, Three Mile Island) actually tells us that events to be classified as "nuclear power plant accidents" take and follow often an unpredictable course. So, while there may be several specific sequences subject to modeling, encoding and application for disturbance analysis, it is at least doubtful whether a cause-consequence model based only on a list of rigidly predetermined sequences of failure modes of plant systems or components could be a useful diagnostic tool in such a complicated case. Also, any attempt to be "complete" in deriving such a list would be dwarfed by the complexity of the systems to be dealt with and the probability of overwhelming number of disturbances which could be possibly affecting them.

One possibility to approach this difficult problem could be to try to find common patterns characterizing groups of similar disturbances which can affect a system, and thereafter design the DAS in such a way as to recognize, upon occurrence of a particular disturbance, the essential features identifying the disturbance as belonging to a particular group. A "group" should accordingly be defined by considering not only the similarities among the single disturbances constituting it, but also the type of (safety) action required of the plant operators in the presence of any of them. In this way the DAS, after identification of the "group" to which an occurring disturbance belongs, could suggest the actions (if any are required) which are proper (in presence of any disturbance belonging to that "group") to maintain the plant in safe conditions.

For the very first step of verifying the validity of this approach, a thorough investigation of many different transient and perturbed conditions in the plant should be performed, to see if such a classification can be attempted, and if indeed certain procedures can be generally followed as a counteraction for a group of disturbances, rather than for a very specific one. Such an investigation clearly requires a powerful simulation tool such as a digital computer code, capable of reproducing the process behavior, including control and safety systems, in the presence of any kind of plant transient, be it a localized system malfunction or a full scale plant accident. If the idea of disturbance classification by "groups" is viable, the same simulation tool will have to be used to provide the information necessary to actually perform it. In any case an accurate and systematic study of the plant behavior in many different upset conditions seems to be a probable starting step to decide which type of approach is required to solve the difficult problem of safety related application of DA systems in nuclear power processes a field that on the other hand is too promising of good rewards in the area of improving further the nuclear plant safety record to be left unexplored.

ANOMALOUS TRANSIENTS

In order to postulate an accident in a logical manner, an initiating event or a top fault event is first defined. The requirement, after an initiating failure, is definition of the functions to be performed and the interrelationship between the various functions. Next the systems provided to perform the functions are identified and interrelationships between the operability states of various systems are defined. Thus the event tree can be regarded as a filter into which is fed all pertinent system informations affecting the course of events following an initial failure and out of which comes only logical functional and system relationships. Each branch of an event tree represents a relevant sequence of possible events, or in other words, indicates the availability or unavailability of systems which influence the course of future events. At some point in time, it may not be clear which system variable has dominant effect on the course of future events, for instance whether actuation of a safety valve is required or not. The computer simulation code may be used to resolve the problem and justify the order of events in the event tree. The failure probabilities for different components are usually available for light water reactors or a fault tree analysis may be required. With the failure probability data and event tree defined, the total probability of occurrence of any branch can be calculated. The failure probability data does not necessarily incorporate human errors such as maintenance errors, which can be included as an uncertainty factor in the calculation of accident probabilities. These event tree branches with their corresponding probabilities provide a basis for transient analysis.

To analyze and interpret plant transients it is necessary that pressure, temperature, and flow histories in different parts of a nuclear reactor system be calculated. This information should subsequently be used to assess the propagation of the accident and to find solutions for mitigation or control of the accident. At the same time efforts should also be made to understand physical phenomena peculiar to these transients.

Currently several codes to predict many types of reactor transients are either available or are being developed. The codes that are available now are: TRAC-P1A, RELAP-5, RELAP-3B, RELAP-4/MOD7, RETRAN and IRT. The codes TRAC-P1A and RELAP-4/MOD7 can handle small breaks, but they have inadequate controls for reactor transients and have excessively long running times. For example to run the Three Mile Island-2 accident up to about 7200 seconds, 47000 CPU's are needed on RELAP-4/MOD7. The RELAP-3B code is adequate for many transients but it cannot handle small breaks and has long running time. The IRT code is a good fast running code for pressurized water reactor (PWR) transients. Its control and trip logics are adequate; however, it cannot handle small pipe breaks and needs improvement in its steam generator modeling.

The RETRAN code is quite good for both pressurized (PWR) and boiling water (BWR) reactor transients. This code is built in a semi-modular fashion and is dynamic dimensioned so that additions and improvements in code can easily be made. In its two phase flow modeling, the RETRAN code assumes the flow to be homogeneous equilibrium and utilizes a dynamic slip model between the two phases. It also includes two phase friction multipliers and flow regime maps. Its heat transfer model is presumably based on local conditions and is a two surface model. Its pressurizer models can handle non-equilibrium conditions. The trip and control logics in RETRAN are quite adequate. However, it cannot handle small breaks and has long running times.

Thus, at the present time none of the available codes possess all the desirable features of a good code for analyzing system transients. Both TRAC-PF1 code for PWR's and RAMONA-III code for BWR's which will be available in the near future would hopefully have improved capability. In this work IRT code was used to analyze PWR system transient due to rupture in steam line. The IRT code was chosen because of its simplicity and fast running time. Next a detailed description of IRT code is given followed by a discussion of results for transient initiated by a steam line break.

Description of Simulation Code IRT

IRT program describes the behavior of a pressurized water reactor during postulated accidents or power transients, such as reactivity excursions, large breaks in primary and secondary coolant systems or pump failure. The program calculates flows, mass and energy inventories, temperatures, primary and secondary loop pressures along with variables associated with reactor power, reactor heat transfer or control systems. Its versatility and efficient computing time allow one to describe simple hydraulic systems as well as complex reactor systems for times on the order of hours after the accident initiation.

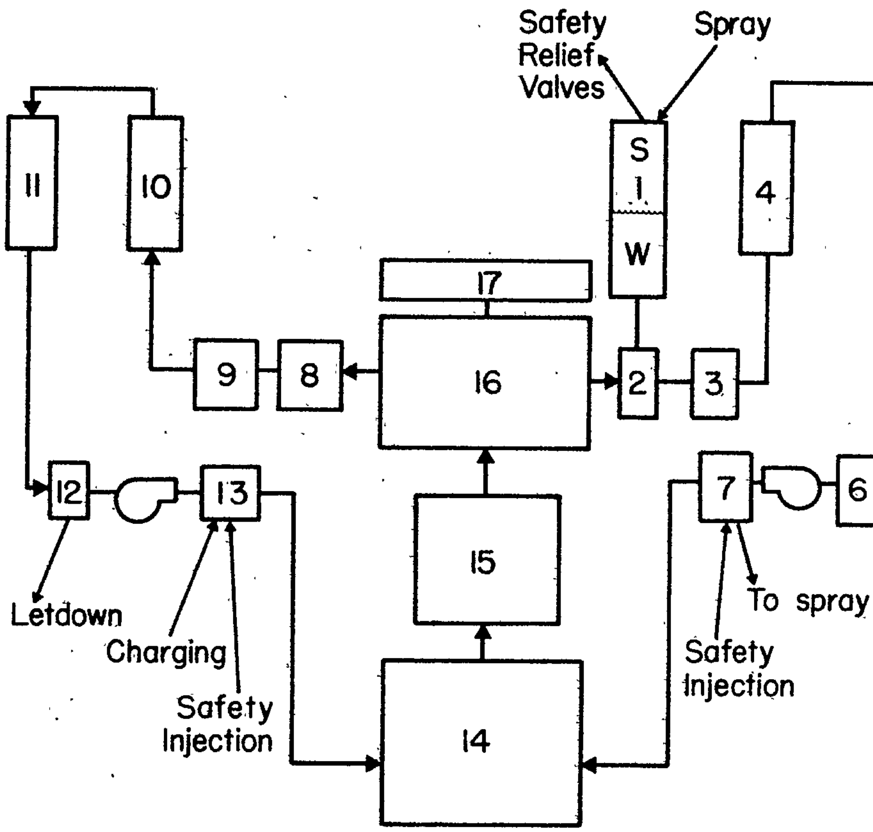
The program is basically designed to analyze PWR transients, however, it can be extended for use in BWR analysis also. Reactor primary thermal hydraulic model (Figure 2) consists of seventeen fixed control volumes including the core, pressurizer and two steam generators in two separate coolant loops. Control volumes are connected by flow paths which may contain a pump. The reactor secondary loops consist of main and auxiliary feedwater, steam generator shellside, and the steam line, up to and including the turbine admission valves. The program basically solves the conservation of energy and mass equations along with the equation of state and the constant volume constraint imposed on each control volume. Thus 50 linear differential equations involving 50 unknowns are to be solved in the primary thermal hydraulics part of the program. In order to reduce the calculation time, the set of 50 simultaneous equations has been reduced analytically to a set of 19 equations and 19 unknowns. Physically 16 of these equations are energy and mass conservation and the other three vary in form depending on the pressurizer status. The system of differential equations is transformed to a set of explicit difference equations by forward differencing and solved by Gaussian elimination. Interfaces between the primary coolant and other nuclear steam supply system components are represented by heat fluxes and external flows. The neutronic model consists of a point kinetic equation with eleven delayed groups. Several engineering safety features are built into the system such as, chemical volume control system, pressurizer spray system, pressurizer relief and safety valves, pressurizer heater, charging and letdown, control and scram rods. A number of alarm and trip signals are available which enable the simulation of a variety of transients.

To simplify the basic differential equations and make the program efficient some assumptions and approximations are made which are discussed below.

a) System pressure is assumed to be spatially uniform throughout the entire primary coolant system. However, the frictional pressure drop is considered for the pressurizer surge line during transients. For the rest of the primary system, although friction results in pressure variation around the reactor coolant system loops, coolant properties evaluated at an average pressure are believed to be adequate for transient analysis. The adequacy of this method has been tested against a more sophisticated thermal-hydraulic code which does account for internodal pressure drops. Since the two phase friction pressure drop is higher than a single phase drop, this approximation appeas to be more appropriate for PWR analysis. This assumption, by eliminating the momentum equation, contributes greatly to the computing efficiency of the program. However, it introduces a major deficiency in pump coastdown analysis. For any pump failure transients a pump flow vs time table must be provided by the user.

b) Each reactor coolant system (RCS) control volume with the exception of the pressurizer is assumed to be homogeneous and in thermodynamic equilibrium, and slip between phases is neglected. Since the RCS nodes consist of coolant undergoing turbulent flow, it is expected that voids will exist as relatively small bubbles entrained in the liquid phase. According to this assumption, small quantities of boiling can be studied in the program. However, as the quantity of boiling increases, one would expect significant deviation from homogeneous flow model.

c) Complete phase separation is assumed to occur in the pressurizer. The pressurizer is subdivided into separate steam and liquid regions. The phases do not have to be in thermal equilibrium. The thermodynamic state of each pressurizer region is calculated with mass and energy transport across the steam-liquid interface. Separation of phases is a reasonable assumption only in quasi-steady state conditions, since the pressurizer fluid is stagnant. The steam bubbles and liquid droplets are assumed to be removed instantaneously. This assumption is made for calculational convenience and has negligible effect on most transient results. However, the swelling of pressurizer liquid level cannot be properly treated by this model. This model would be highly erroneous in case of depressurization of the pressurizer (i.e. relief valve stuck open) if the level swells high enough to cause two phase flow through the valve. Surface evaporation and condensation at the steam liquid interface have been neglected as the associated mass and energy transfer rates are expected to be small.



Node	Physical Description
1	Pressurizer
S	Steam region in pressurizer (if present)
W	Liquid region in pressurizer (if present)
2	Upstream half of hot legs
3	Downstream half of hot legs
4	Steam generator inlet plenum plus upstream half of tubes
5	Steam generator outlet plenum plus downstream half of tubes
6	Upstream half of cold legs
7	Downstream half of cold legs
8	Same as 2 in other steam generator loop
9	Same as 3 in other steam generator loop
10	Same as 4 in other steam generator loop
11	Same as 5 in other steam generator loop
12	Same as 6 in other steam generator loop
13	Same as 7 in other steam generator loop
14	Reactor vessel inlet plenum
15	Core
16	Reactor vessel outlet plenum

Fig.2 Primary System Model

d) The spray flow is assumed to be 100% effective in condensing steam. Since the pressurizer spray nozzle breaks the spray flow into droplets of sufficiently small size and, consequently large surface area per unit of liquid mass, this assumption appears to be reasonable.

One of the most important characteristics of IRT code is its efficient computing time. This makes it possible to study transients which may take relatively long periods of time (on the order of an hour) to resolve. High efficiency has been achieved at the cost of making some approximations which limit the capabilities of the code as explained above. The lack of a momentum equation in the code can be corrected by incorporating an integral form of momentum equation to balance forces around the loops without reducing the codes efficiency to unacceptable levels. To treat significant boiling of the primary coolant in a more appropriate way, a two phase slip model needs to be added to the code. The code at the present time does not have independent restarting capabilities, which can be programmed without any reduction in efficiency. Finally, to predict the pressurizer level with better accuracy a bubble rise model should be incorporated in the pressurizer calculations.

Addition of the models described above will substantially increase the capabilities of the code, however, there is a tradeoff between the code's capabilities and its efficiency.

Simulation of a Steam Line Break

To demonstrate the use of the simulation code in transient analysis a steam line rupture in a PWR secondary loop is described for a typical Combustion Engineering, 2560 MWt, two loop plant. The case studied falls in the general category of, "The Effects of Secondary System Failures on the Primary System". As a first step in the analysis of the transient following the initial failure, an event tree can be drawn to express the system availabilities and functional interrelationships [1]. Figure 3 shows a representative event tree. The initiating or transient event (TE), can be a steam line break, or a steam line break concurrent with steam generator tube rupture. The column headings represent the following pertinent systems.

Reactor Scram: A rapid insertion of control rods makes the reactor subcritical within seconds. However, because of the cooldown of the primary system due to steam generator depressurization, the reactor may return to critical state. Failure to scram results in an ATWS event which may be discussed in another event tree.

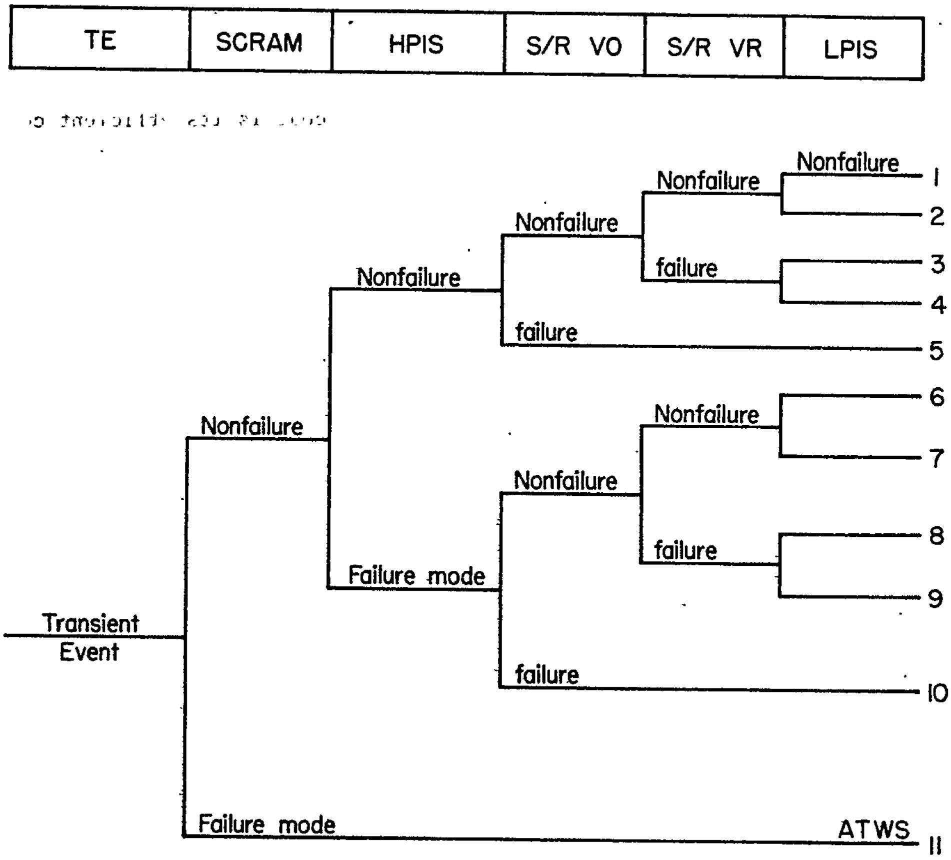


Fig.3 PWR Transient Event Tree for Steam Line Break

High Pressure Injection System (HPIS): High pressure injection mode of the chemical volume control system injects concentrated boron solution in the hot leg to return the core to subcritical state.

RCS Safety/Relief Valves Open (S/R VO): This column heading represents the opening of the reactor coolant system pressurizer relief and safety valves upon demand.

Safety/Relief Valves Reclose (S/R VR): The RCS pressurizer safety and relief valves that open as a result of a transient event must reclose to prevent a discharge of excessive quantity of coolant. Otherwise a valve staying open following a transient results in a LOCA.

Low Pressure Injection System (LPIS); Low pressure injection mode of safety injection system will continue to inject borate water into the system, and also help cool this core during later stages of the transient.

The sequence No. 1 of the event tree in Figure 3, with the initiating event (TE) being the steam line break is described below.

Failure of lines in the PWR secondary coolant systems have occurred principally during plant testing and start-up periods. These types of failures have included inadequate initial design of relief valve headers in the steam supply lines, discharge of secondary coolant from leaking feedwater valves, discharge of secondary coolant from cracks in main feedwater lines. The steam line break transient analyzed is a full shear rupture of the main steam line causing a rapid depressurization of all steam generators. This depressurization and consequent reduction in secondary temperature results in increased primary to secondary heat transfer. This, in turn, lowers the primary system temperature and pressure and increases the moderator density. The increased moderator density adds reactivity due to the negative moderator feedback coefficient and gives a potential for a recriticality and core power excursion. The initial conditions for this analysis correspond to hot zero power (HZP) conditions. The HZP conditions were selected since these conditions produce a greater potential for a recriticality following the steam line break based on the results presented in a typical CE FSAR. The results of the transient analysis show that steam generator pressure drops to the scram setpoint (478 psi) in 3.9 seconds. Following a 0.9 second delay, the control rods are inserted, turbine trip is initiated, and the main steam isolation valves begin to close isolating the intact steam generator. Primary pressure drops rapidly (Fig. 4) as pressurizer liquid rushes into the hot-leg piping, pressurizer empties completely in 10.5 seconds and the upper head region flashes at 13.97 seconds at which time primary system pressure has reduced to saturation pressure corresponding to the upper head temperature. The upper head region is now controlling the primary system pressure. This continues until the upper head liquid is depleted (greater than 50% quality), and additional primary system nodes begin to flash (cold-leg piping, hot-leg piping, and outlet plenum). The reactor becomes critical in 41.8 seconds and a power transient is initiated. Safety injection water reaches the primary system at 42.3 seconds; however, this water is initially unborated and does not terminate the power transient. The power transient is eventually terminated by the boron injection with both the safety injections and accumulator water. The peak core power and heat flux are 30% and 25% rated, respectively. Figure 5 shows the total reactivity variation with time.

Accidents and anomalous transients can be broken down into three groups with respect to their length of duration. Large break loss of coolant accidents with severe consequences in the early stages of transient fall in the first category. Next, come operational transients which may have less severe consequences. The most unsteady part of these transients may last quite long, thus

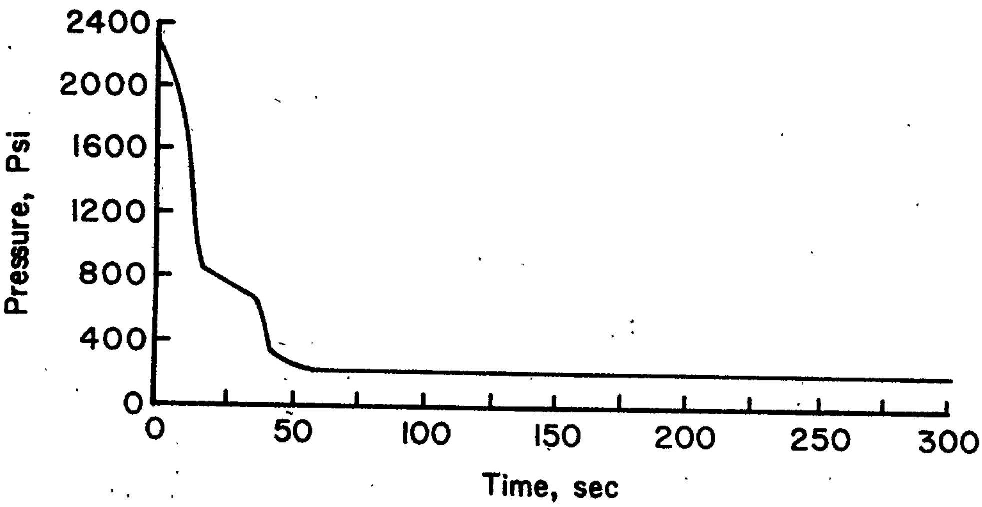


Fig.4 Primary Pressure as a Function of Time

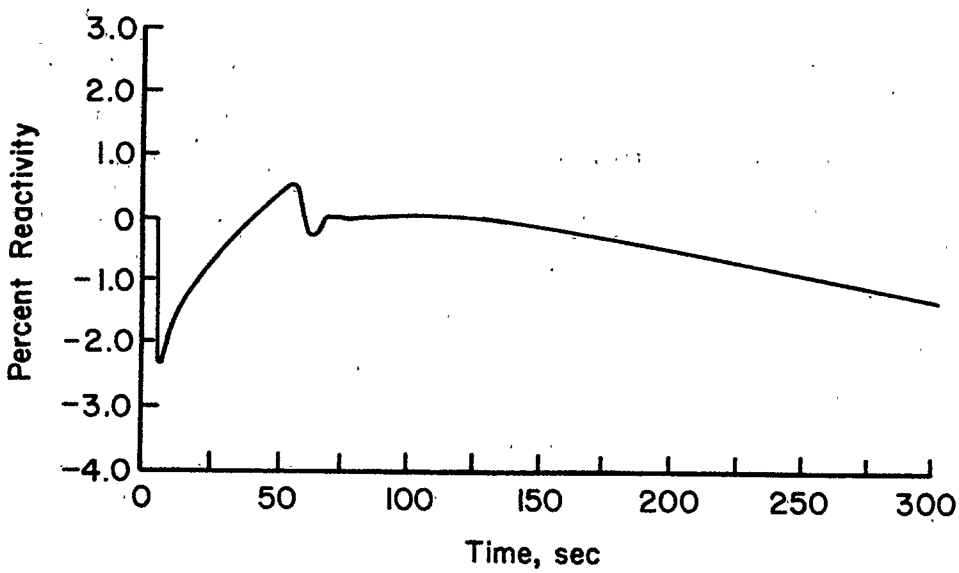


Fig.5 Total Reactivity as a Function of Time

requiring longer period of time (of the order of an hour). The third type of transient occurs at shut down conditions with very low flow rates (possibly dominated by free convection) and low heat generation rates. This type of transient may take several days to resolve.

The steam line break transient analyzed above shows some of the capabilities of the simulation code IRT which is designed for operational transients for times on the order of an hour. It is necessary that all pertinent related safety systems such as relief and safety valves, safety injection system be modeled in the simulation code. If able to comply with these and other requirements a code should be capable of analyzing the transient at least during its most unsteady part, which in the case discussed, was about 300 seconds.

CONCLUSIONS

Disturbance Analysis System (DAS) is shown to have potential application in improving plant availability and plant safety. However, at present our understanding as well as capability in employing DAS for complex systems such as nuclear steam supply systems is limited. Considerable work needs to be done in this area. Presently available transient system codes are not capable of handling all types of operational transients. Also, some of these codes have long running times and are weak in modeling and control and trip logics. Future efforts should be made to improve these codes and to provide online guidance to operators in case of anomalous transients.

ACKNOWLEDGMENT

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